D. R. Madison (Dennis)
Vice President - Hatch

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SOUTHERN COMPANY
Energy to Serve Your World℠

October 5, 2007

Docket No.: 50-366 NL-07-1820

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report
Reactor Scram on Low Water Level due to Partial Loss of Condensate System

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a partial loss of the condensate system which resulted in an automatic reactor scram and primary containment isolation actuation.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Dennis Madison

D. R. Madison
Vice President – Hatch

DRM/OCV/daj

Enclosure: LER 2-2007-008

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. R. Madison., Vice President – Hatch
Mr. D. H. Jones, Vice President – Engineering
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Hatch
Mr. J. A. Hickey, Senior Resident Inspector – Hatch

NRC FORM 366 (9-2007)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Reactor Scram On Low Reactor Water Level Due to Partial Loss Of Condensate System

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
08	07	2007	2007	- 008 -	00	10	05	2007	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)							
		20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
		20.2201(d)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
		20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
		20.2203(a)(2)(i)		50.36(c)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
10. POWER LEVEL: 100		20.2203(a)(2)(ii)		50.36(c)(1)(ii)(A)	X	50.73(a)(2)(iv)(A)		50.73(a)(2)(x)
		20-2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
		20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
		20.2203(a)(2)(v)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(C)		OTHER
		20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Edwin I. Hatch / Kathy Underwood, Performance Analysis Supervisor	912-537-5931

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (*If yes, complete 15. EXPECTED SUBMISSION DATE*)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On August 7, 2007 at 15:06 EDT, Unit 2 was at 100 percent power in Mode 1 at 2804 CMWT. At this time a partial loss of the Condensate System caused low Feedwater flow resulting in a Reactor Protection System (RPS) actuation on Low Reactor Water Level and a Group 2 Primary Containment Isolation System (PCIS) isolation. The investigation determined that the loss of condensate system was caused by the loss of the 2D 4160V switchgear. The 2D bus provides power to the 2A and 2B Condensate and Condensate Booster Pumps. The loss of the 2D 4160V bus was the result of the inadvertent actuation of the over-current relay protecting one of the three phases of the normal supply breaker. This relay had been removed from service as part of a routine I&C periodic relay calibration. During reinstallation of the relay cover following completion of the calibration, the relay actuated tripping the normal supply breaker for the 2D 4160V switchgear and de-energizing the bus providing power to the pumps. All systems functioned per their design given the water level transient.

The root cause of this event was determined to be ineffective execution of a screening procedure written to determine scram/transient potential of I&C activities. The screening procedure was executed for the calibration of the overcurrent relay and errantly determined that there was no reactor trip potential when performing the procedure on-line and did not include a precaution for installation of relay cover. The procedure has been revised to correct these inadequacies and all I&C procedures previously screened will be re-evaluated.

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NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

This event is reportable, per 10 CFR 50.73 (a)(2)(iv)(A), because an event occurred which resulted in an automatic scram and automatic closure of primary containment isolation system valves.

On August 7, 2007 at 15:06 EDT, Unit 2 was in Mode 1 at 2804 CMWT, 100 percent power. An inadvertent actuation of contacts on an over-current relay protecting one of the three phases for the normal supply breaker of the 2D 4160V switchgear occurred. This resulted in a partial loss of the Condensate System (EIIS Code SD) causing a reduction in Feedwater (EIIS Code SJ) flow. The reduction in Feedwater flow caused a decrease in reactor water level. The Recirculation System (PCIS, EIIS Code AD) runback setpoint was reached and the runback initiated. With the loss of two of the three condensate pumps the Recirculation System runback was not able to prevent the continued decrease in reactor water level. As water level continued to decrease, the Reactor Protection System (RPS, EIIS Code JC) reached the reactor low water setpoint and initiated an automatic scram signal. The Group 2 Primary Containment Isolation System (PCIS, EIIS Code JM) actuation setpoint was reached and the isolation signal initiated. Following the reactor scram, water level was recovered automatically with the normal condensate and feedwater system. No automatic initiation setpoints were reached for the Emergency Core Cooling Systems and the operators had no need to manually actuate those systems.

CAUSE OF EVENT

The root cause of this event was determined to be ineffective execution of a screening procedure written to determine scram/transient potential of I&C activities. The screening procedure was executed for the calibration of the overcurrent relay and errantly determined that there was no reactor trip potential when performing the procedure on-line and did not include a precaution for installation of the relay cover.

SAFETY ASSESSMENT

Following the automatic scram on low reactor water level, reactor vessel water level continued to decrease due to void collapse. Level reached a minimum of about thirty two inches below instrument zero (about 126 inches above the top of the active fuel). The decrease in water level resulted in a Group 2 PCIS isolation on low water level and thus closure of the Group 2 Primary Containment Isolation Valves per design. The RPS and PCIS are Engineered Safety Feature systems.

The operating Reactor Feedwater Pumps automatically restored water to its designed setpoint. Operations personnel verified correct system response and restored the isolation valves and the RPS to their normal configuration.

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NARRATIVE

All systems functioned as expected and per their design given the water level transient. Water level was maintained well above the top of the active fuel throughout the transient and was restored to its desired value without the need for emergency core cooling system actuation. Therefore, it is concluded the event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Procedure 57CP-CAL-108-2, "Westinghouse CO Over Current Relay," has been revised to delete the statement indicating "no trip potential" and precautions for installation of relay covers have been added.

Procedure AG-MNT-03-0606, Procedure Review for Trip Potentials, will be revised to include guidance to assume components affected (or potentially affected) by the activity are actuated during the activity for the determination of a trip and transient potential. Implementation of this corrective action will be tracked under the corrective action program.

All I&C procedures previously screened will be re-evaluated per the revised guidance. Implementation of this corrective action will be tracked under the corrective action program.

ADDITIONAL INFORMATION

Other Systems Affected: None

Failed Components Information: None

Commitment Information: This report does not create any permanent licensing commitments.

Previous Similar Events:

LER 2-2006-002 identified an instance where performance of a calibration procedure resulted in an automatic reactor scram. The root cause of that event identified an error in the calibration procedure which allowed work to be performed on-line that would directly cause an automatic scram if performed on-line. Additional corrective actions for that event required a review of calibration procedures to determine if an automatic scram would be a potential effect of performing the procedure on-line. This review failed to achieve the desired result of identifying procedures which could result in a unit scram if performed on-line. Therefore the corrective action for the event reported in LER 2-2006-002 was not effective in preventing the current event.